ACCESSION #: 9906040173

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: MONTICELLO NUCLEAR GENERATING PLANT PAGE: 1 OF 6

DOCKET NUMBER: 05000263

TITLE: Feedwater Controller Power Supply Failure Causes Low

Reactor Water Level Scram and Group 2 & 3 Isolations;

Subsequent Events Cause HPCI to become Inoperable

EVENT DATE: 04/22/99 LER #: 99-004-00 REPORT DATE: 05/24/99

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

50.73(a)(2)(v)

LICENSEE CONTACT FOR THIS LER:

NAME: Arne Myrabo TELEPHONE: (612) 295-1266

COMPONENT FAILURE DESCRIPTION:

CAUSE: X SYSTEM: JB COMPONENT: JX MANUFACTURER:

REPORTABLE EPIX: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

Failure of a digital feedwater control system (DFCS) power supply caused one feedwater regulating valve to close and the other to lock-up in its pre-existing condition. Attempts

to manually control reactor water level failed due to the DFCS failure, resulting in an automatic reactor low water level scram. Subsequently, steam lines for the main steam system, High Pressure Coolant Injection system (HPCI) and Reactor Core Isolation Cooling system (RCIC) were overfilled, causing HPCI and RCIC to become inoperable for about 1/2 hour. Water level was restored to shutdown normal conditions approximately one hour later.

The DFCS power supply failure was due to an oxidized connection in a plus 5 volt power supply. The overfill was caused by the failed level indications and misleading Safety Parameter Display System (SPDS) indications. Concern about using reactor water level indications led to bypassing the Reactor Feedwater Pump (RFP) high water level trip.

Three power supplies were replaced and all associated connections cleaned. Several operating procedures were revised, temporary monitoring instrumentation was added and extensive testing was performed. Operator training was provided on several topics.

Conditions Prior to the Event

On April 12, 1999 two DFCS modules locked up. The failure mode did not affect actual water level and no plant transient occurred. Troubleshooting efforts focused on the two modules and resulted in replacing one module and resetting the other module. During the troubleshooting, no diagnostic flags were found to indicate any concern with power supplies. All testing after repairing the modules showed functions to be normal. With the plant on-line, the extent of troubleshooting was limited and did not allow for testing the power supplies as was done in the April 22, 1999 event.

Description

At 1:30 AM on April 22, 1999 while operating at 100% power, a plus 5 volt DFCS 1_/ power supply became erratic due to an oxidized connection. This caused erratic operation of many of the DFCS modules connected to the plus 5 volt bus even though there are redundant power supplies supplying the same plus 5 volt bus. The "B" feedwater regulating valve closed and the "A" locked up in its pre-existing condition. NOTE: the "B" feedwater regulating valve partially re-opened a short time later. One feedwater flow indicator failed as-is at the 100 percent power flow value and the other followed the event. The "B" and "D" main steam line flow indicators failed as-is at the 100 percent power flow value and the others followed the event. One feedwater reactor water level indicator was reading a constant 3 inch and the other followed the event.

Attempts to manually control reactor water level failed due to the DFCS failure. The reactor scrammed at 1:31 AM on low water level; a Group 2 and Group 3 Isolation occurred as expected at the low water level point.

As a result of the abnormal level indicator, the crew was concerned with reactor water level accuracy following the scram. While water level was rising, the operator bypassed the RFP High Reactor Water Level Trip. This prevented the feedwater pump trip at plus 48 inches. There was no specific procedural guidance on the use of the RFP High Reactor Water Level bypass switch, other than for RFP testing during outages.

When indicated reactor water level exceeded 60 inches, the panel indicators for reactor water level showed the safeguards level offscale high, one feedwater level offscale high and the other reading 3 inches.

As level was increasing, the SPDS level validation screen was used to validate reactor water level indication. Note that the DFCS failure did not affect the SPIDS feedwater instrument readings, since the SPDS level input was not compromised (not associated with the DFCS). SPDS showed 3 valid

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1_/ EIIS Component Code: JX, EIIS System Code: JB

level instruments: 2 feedwater and the vessel floodup reactor water level instruments. At this time, water level was above the feedwater reference leg taps such that they were no longer responding to actual level changes. However, the reactor vessel floodup level instrument was responding to level changes.

SPDS was programmed to output valid feedwater levels up to 95 inches reactor vessel level. In this case the reference leg tap was covered and the instruments indicated approximately 80 inches. Since there were two feedwater "valid" instruments with one reactor vessel floodup level indication, SPDS "voted" and produced a validated level of 80 inches. This indicated to the operators that level was stable at 80 inches when in fact it was increasing and eventually covered the main steam line nozzles.

The B Feedwater Regulating valve was closed 9 seconds following reactor level reaching plus 48 inches. The feedwater block valves were subsequently closed to terminate the feedwater injection from the A Feedwater Regulating valve, which was locked in a partially open position.

Following the reactor scram, reactor water level rose to approximately 150 inches. The bottom of the main steam lines is at 109 inches; as a result, water entered the main steam lines. This condition caused high drain pot level alarms on the RCIC and HPCI systems.

HPCI was declared inoperable at this time. The condition was cleared approximately 1/2 hour later. About 24 hours later, HPCI was again declared inoperable for approximately 5 minutes (per procedure) when the high drain pot level alarm was received while cycling the HPCI steam supply valve to prevent thermal binding.

Water level was reduced to below the steam lines within 1/2 hour and restored to shutdown normal conditions approximately one hour after the scram.

Cause

The cause of the reactor scram was the DFCS power supply failure. The main steam line overfill was caused by the failed level indications and misleading Safety Parameter Display System (SPDS) indications. Concern about the validity of the reactor water level indications led to bypassing the Reactor Feedwater Pump (RFP) high water level trip, which prevented the RFPs from tripping.

Analysis of Reportability

This report is being submitted per 10 CFR 50.73(a)(2)(iv), since an automatic actuation of the reactor protection system occurred that was not part of a pre-planned sequence, and 10 CFR 50.73(a)(2)(v), since the steam line overfill caused HPCI to be inoperable for approximately 1/2 hour on one occasion and 5 minutes on a second occasion.

Safety Significance

Post-scram recovery occurred without further complication.

Although level was indicated to be above the steam lines, pressure did not increase, nor were auxiliary systems relying on main steam affected (recombiners, air ejectors and the steam sealing system stayed in service). It is believed that a steam and water mixture entered the steam lines and was removed through the turbine bypass valves to the condenser.

The main steam lines to the Main Steam Isolation Valves (MSIV) have previously been analyzed with the assumption they are filled with water. During the reactor coolant pressure boundary leak test these lines are filled with water. Field inspections were performed of the main steam lines and other lines that could have been filled with water; no adverse effects were noted. The main steam lines past the MSIVs up to the main turbine stop valves are analyzed to USAR Chapter 12 Class I requirements (without the assumption of being filled with water). The main steam lines beyond the main turbine stop valves are not affected since water was not in these lines (main turbine stop valves were closed). The main steam lines downstream of the MSIVs and the HPCI and RCIC lines to their respective turbines were reviewed by calculation and determined not to have been overstressed with the added water.

HPCI, RCIC and the SRVs were out of service for a short time; the reactor was placed in a cold shutdown condition within the 24 hours required by technical specifications for this condition.

Therefore, the health and safety of the public were not affected by this condition.

Actions Prior to Startup

Three DFCS power supplies have been replaced and the oxidized connection was thoroughly cleaned, along with similar connections.

Preventive maintenance has been performed on the DFCS (load test and calibration).

Temporary instrumentation has been installed to provide on-line monitoring of the power supplies, which will be checked on a daily basis.

The SPDS calculation for the feedwater reactor level instrument was changed to remedy the error where valid level is shown above the instrument taps. In light of this event, an independent review of the other SPDS reactor water level displays has been performed to assure that no similar calculation problems exist.

Procedural changes have been made to address the use of the RFP Reactor high water level trip bypass switch and clarify operator actions for:

Condensate and Reactor Feedwater System Operation

Startup Procedure and Shutdown Procedure

Loss of Reactor Water Level Control Procedure

Alarm Response Procedure FW CONTROL HARDWARE TRBL and FW CONTROL SIGNAL FAILURE

Management expectations with respect to operator responses to transients and the use of bypasses for design features have been conveyed to operations personnel.

An operator aid for reactor vessel floodup instrument temperature compensation has been issued.

Follow-up Actions

Training and procedures will be reviewed to ensure that management expectations on operator response to transients and the use of bypasses of design features are reinforced. Training on identification of control system failures will be evaluated and operator actions that are not specifically directed in procedures will be identified.

A root cause determination of the use of the RFP high reactor water level trip bypass switch will be performed.

The DFCS will be replaced. Lessons learned from this event will be considered in the design of the new DFCS (including root cause of the power supply failure and possible on-line diagnostics) and improvements to the PM program will be considered.

SPDS - The software change in the simulator will be installed in the simulator. Enhanced display for SPDS reactor water level instrumentation will be considered. The SPDS logic, which does not display the compensated reactor vessel floodup water level when it's

the only "valid" instrument available will be evaluated.

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Failed Component Identification

Digital Feedwater Control System 5 volt power supply: AUTECH, Model

VT100

Similar Events

None.

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NSP Northern States Power Company

Monticello Nuclear Generating Plant

2807 West Hwy 75

Monticello, Minnesota 55362-9637

May 24,1999 10 CFR Part 50

Section 50.73

US Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50-263 License No. DPR-22

LER 99-004

Feedwater Controller Power Supply Failure Causes

Low Reactor Water Level Scram and Group 2 & 3 Isolations;

Subsequent Events Cause HPCI to Become Inoperable

The Licensee Event Report for this occurrence is attached. This report contains no new NRC commitments.

Please contact Arne Myrabo at (612) 295-1266 if you require further information.

Byron Day

Plant Manager

Monticello Nuclear Generating Plant

c: Regional Administrator - III NRC

NRR Project Manager, NRC Attachment

Sr Resident Inspector, NRC

State of Minnesota, Attn: Steve Minn

Attachment

*** END OF DOCUMENT ***